Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
PO Box 249
Buchanan, NY 10511-0249
Tel 914 734 5340
Fax 914 734 5718

Fred Dacimo
Vice President, Operations

March 11, 2003
NL-03-043

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Indian Point Nuclear Power Plant Unit 3
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 2003-001-00

Dear Sir:

The attached Licensee Event Report (LER) 2003-001-00 is hereby submitted in accordance with the requirements of 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73(a)(2)(iv) for an event recorded in Entergy's corrective action process as Condition Report CR-IP3-2003-00160.

Entergy is making no new commitments in this LER. Should you have any questions regarding this submittal, please contact Mr. John McCann, Manager of Licensing, Indian Point Energy Center at (914) 734-5074.

Very truly yours,

Fred R. Dacimo
Vice President, Operations
Indian Point Energy Center

cc: See next page

IE22

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406-1415

Mr. Patrick D. Milano, Senior Project Manager
Project Directorate I
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

INPO Record Center
700 Galleria Parkway
Atlanta, Georgia 30339-5957

U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant
P.O. Box 337
Buchanan, NY 10511-0038

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Indian Point Unit 3	05000- 286	1 OF 5

4. TITLE

Manual Reactor Trip Due to High Differential Pressure Between Condenser Sections

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000-
1	13	2003	2003	– 001 –	00	03	11	2003	FACILITY NAME	DOCKET NUMBER 05000-

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20 2201(b)		20 2203(a)(3)(ii)		50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50.73(a)(2)(iii)	50 73(a)(2)(x)
	20 2203(a)(1)		50 36©(1)(i)(A)	X	50.73(a)(2)(iv)(A)	73 71(a)(4)
	20 2203(a)(2)(i)		50 36©(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
	20 2203(a)(2)(ii)		50 36©(2)		50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	20 2203(a)(2)(iii)		50 46(a)(3)(ii)		50.73(a)(2)(v)(C)	
	20 2203(a)(2)(iv)		50 73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
	20 2203(a)(2)(v)		50 73(a)(2)(i)(B)		50.73(a)(2)(vii)	
	20 2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)	
	20 2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
John Ventosa, System Engineering Manager	(914) 736-5224

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	KE	MO	G080	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i e , approximately 15 single-spaced typewritten lines)

On January 13, Operations manually tripped the reactor in accordance with Off Normal Operating Procedure ONOP-C-1 due to high differential pressure (dp) between sections of the main condenser. The high dp was due to a loss of circulating water in one condenser section as a result of the trip of the 35 circulating water pump (CWP) while the 36 CWP was tagged out of service for planned maintenance. All control rods fully inserted. The plant was stabilized in hot standby with decay heat being removed by the main condenser. Offsite power remained available and the emergency diesel generators did not start. The auxiliary feed water (AFW) system automatically started. The cause of the event was a high dp between condenser sections. The apparent cause of the 35 CWP trip was a failure of the positive DC exciter lead terminal lug that connects the exciter rotor to the main rotor of the pump motor. The failure was a result of the lead rubbing against the motor dust cover due to improper cable position during a previous maintenance activity. Corrective actions included troubleshooting and repair of the 35 CWP, testing and repair of the CWP Standby drive, and return to service of the 36 CWP. A discussion of the event and management expectations on human performance was communicated to the maintenance department and site population. Maintenance procedure MTR-004-CWP was revised to provide a caution regarding motor exciter leads and assembly details. Procedure PMP-052-CWS will be revised to include a requirement to open and inspect CWP motors brought on site from a vendor. The event had no effect on public health and safety.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within brackets { }

DESCRIPTION OF EVENT

On January 13, at approximately 0618 hours, while at 100% steady state reactor power, Operations manually tripped {JC} the reactor {RCT} in accordance with Off Normal Operating Procedure ONOP-C-1 due to greater than three (3) inches differential pressure (dp) between sections of the main condenser {SG}. The high dp was due to the trip of the 35 Circulating Water {KE} Pump {P} (CWP) while the 36 CWP was tagged out of service for planned maintenance. The 35 and 36 CWPs supply one section of the three-section condenser. The loss of both pumps in the same condenser section caused a partial loss of vacuum for that section resulting in a high dp between adjacent condenser sections.

On January 13, at approximately 0608 hours, the 35 CWP tripped from its normal Load Commutated Inverter (LCI) drive and transferred to the standby LCI drive. Within approximately 15 seconds of the transfer, the 35 CWP tripped from the standby drive. Central Control Room (CCR) {NA} operators observed an indicator light for the 35 CWP normal breaker open and then the 35 CWP standby light come on after a few seconds followed by a condenser low vacuum alarm. Operations dispatched operators and Instrument & Control (I&C) personnel to investigate the condition. Dispatched operators reported to the CCR that they reset all fault lights locally and attempts to restart CWP 35 were unsuccessful. At approximately 0615 hours, CCR operators entered ONOP-C-1 and manually tripped the reactor at approximately 0618 hours, in accordance with the procedure, for exceeding three inches differential pressure between condenser sections.

CCR operators observed the rod bottom lights, Reactor Trip (RT) First Out Annunciator (Manual Trip), and Turbine Trip First Out Annunciator (Reactor Trip). CCR operators entered Emergency Operating Procedure (EOP) E-0, Reactor Trip or Safety Injection, then ES-0.1, Reactor Trip Response, and transitioned to Plant Operating Procedure (POP) 3.1, Plant Shutdown from 45% Power. The plant was stabilized in hot standby with decay heat being released to the main condenser via the steam dump valves {V} and the transient terminated. All control rods {AA} fully inserted. Station offsite power remained available and there was no automatic start of the emergency diesels {EK}. The Auxiliary Feed Water (AFW) system {BA} automatically started as expected due to changes in Steam Generator level from full power operation. The following systems failed to function properly; 1) the 32 Reactor Coolant {AB} Pump {P} tripped when the feed to its supply bus auto transferred from the Unit Auxiliary Transformer to the Station Auxiliary Transformer {XFMR}, 2) the 32 source range detector {IG} did not indicate as required after it energized (read low), 3) the Plant Vent Gas radiation monitor {IL} R-14 alarmed and spiked then returned to normal, and 4) CCR received an AFW low flow alarm requiring an operator to manually open the 31 AFW pump recirculation valve in accordance with Alarm Response Procedure (ARP-006).

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

At 0847 hours, a four hour non-emergency notification (Incident Log No. 35506) was made to the NRC for a Reactor Protection System (RPS) actuation in accordance with 10 CFR 50.72(b)(2)(iv)(B). Operations recorded the event in the corrective action program (CAP) as Condition Report CR-IP3-2003-00160. A post transient evaluation (Report No. 03-01) was completed on January 14, 2003.

The CWPs are motor-driven variable speed type pumps {P} manufactured by Allis-Chalmers (A180}, Each CW pump drive unit consists of a variable speed synchronous motor {MO} powered from an adjustable-frequency power control system of the LCI (INVT) type. The motor rotor (field) is excited by an exciter (DC generator) coupled to the motor shaft. Excitation is controlled by the excitation voltage controller (EVC) {EC}. Controlled excitation is necessary for load commutation. The exciter's rotor voltage is rectified by a 3 phase full wave rectifier on the rotor to supply DC field current to the motor. The CWP motors and LCI drives were manufactured by General Electric (G080). Six LCIs are dedicated to serving individual CW pump motors. A seventh LCI is a spare that is used as a standby drive and can replace any failed dedicated LCI.

A detailed engineering evaluation was performed to determine the cause of the 32 RCP trip. The RCP breaker trip was initiated by an overcurrent protection relay trip however no failure mechanism was identified. RCP motor testing and assessment of motor currents showed no anomalies. Motor feeder cable testing results were satisfactory. The motor protective relay was checked for damage or miscalibration and found satisfactory. The RCP bus tie breaker (UT4-ST6) contacts and arc chutes were checked and no indications of excessive currents were identified. Inspection of the 6.9 KV breaker for the 32 RCP did not show signs of arcing, overheating, or degradation. The evaluation determined that the 32 RCP and associated components are operating as designed and are ready for service.

Instrument & Control technicians performed troubleshooting on the 32 source range detector (N-32) {DET} of the excore nuclear instrumentation system (NIS) {IG} and could not identify the cause of the detector's improper readings. The NIS was manufactured by Westinghouse Electric Corporation {W121}. The detector began tracking N-31 and was within its testing frequency and was returned to service.

CAUSE OF EVENT

The cause of the manual reactor scram was a high dp between condenser sections. The high dp in one section of the three section condenser was due to the loss of both CWPs in one condenser section. The loss of two CWPs was due to the trip of the 35 CWP while the 36 CWP was tagged out for planned maintenance. The apparent cause of the 35 CWP trip was a failure of the DC exciter lead that connects the exciter rotor to the main rotor of the pump motor. The failure was a result of the positive DC lead rubbing the motor dust cover due to improper cable position during a previous maintenance activity. The lead was installed improperly during previous maintenance that installed a new CWP motor upper oil reservoir cooling coil that cools the upper motor bearing lubricating oil. The manufacturer, GE provided instructions on coil replacement but did not provide specific termination instructions and maintenance failed to request them. Subsequently, during coil replacement for the 35 CWP, inadequate clearance was provided for the DC lead that connects the exciter rotor to the main rotor. The DC lead rubbed during motor operation and vibrated the connection to the lug until the connection failed.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the CAP to address the causes of this event and prevent recurrence.

1. Troubleshooting was performed on the CWP LCI Standby drive. One silicon controlled rectifier (SCR) was found shorted in the EVC cabinet and was replaced. The shorted SCR was attributed to the damaged motor exciter wire. One power supply card (NPSE) in the standby drive ECV cabinet was replaced as a predictive maintenance action. Functional testing of the Standby drive was performed and test results were satisfactory. The 36 CWP was transferred to the standby drive and the standby drive was determined to be operational on January 14, 2003. The 36 CWP was transferred to its normal drive and returned to service.
2. Troubleshooting was performed on the 35 CWP motor exciter leads and associated lugs. The positive DC lead and associated lug was repaired. The CWP was tested and returned to service on January 14, 2003.
3. Immediately after discovery, a tailgate meeting was held for the Indian Point 3 Maintenance Department to discuss the event and reinforce management expectations on human performance.
4. A memorandum was issued to the Indian Point Energy Center (IPEC) population to describe the causes of the event and reinforce management's expectation for attention to detail.
5. Maintenance procedure MTR-004-CWP was revised to include a caution about motor exciter rotor leads and necessary assembly details to prevent recurrence.
6. Procedure PMP-052-CWS will be revised to include a requirement to inspect all CWP motors brought on site following any refurbishment completed by an outside vendor to ensure proper assembly of exciter leads.

EVENT REPORTING

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply include the reactor protection system (RPS) including reactor scram or RT, and AFW.

This event meets the reporting criteria because the RPS was manually actuated and a RT occurred. In response to the RT, the AFWS actuated due to steam generator level changes, which occur after a RT from full power.

PAST SIMILAR EVENTS

A review of Licensee Event Reports (LERs) for the past two years did not identify any events that involved a RT caused by high differential pressure in the condenser.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 3	05000-286	2003	- 001	- 00	5 of 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required safety systems performed as designed when the RT occurred. The AFWS actuation was expected due to steam generator level changes, which occur after a RT from high power levels.

There were no significant potential safety consequences of this event under reasonable and credible alternative conditions. A loss of a CWP (e.g., 35 CWP) and therefore cooling to a condenser section may result in a loss of condenser vacuum, loss of megawatts, or high turbine exhaust hood temperatures. Low condenser vacuum will result in a turbine trip. When the unit load is greater than the Permissive P-8 setpoint, a trip of the turbine generator initiates a RT. A loss of external electrical load/turbine trip is an analyzed event described in FSAR Chapter 14. The plant performed as expected and the event was bounded by the FSAR analysis. The trip of the 32 RCP, after transfer of the 32 RCP's normal 6.9 KV power source from Bus 4 to Bus 6, resulted in loss of forced flow in reactor coolant loop 32. The loss of forced RC flow caused by the loss of one out of four RCPs from full power is an analyzed event in FSAR Section 14.1.6. Protection from a partial loss of flow event is provided by a RT. Below the P-7 Permissive, natural circulation flow provides adequate cooling. Following RT, the affected RCP will continue to coast down and a stable plant condition will be attained. The plant performed as expected and the 32 RCP trip event was bounded by the FSAR analysis. For this event rod control was in automatic and the reactor scrammed immediately upon a manual RT. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.